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AUTH.NAME AUTHOR AFFILIATION
MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
VISSING,G.S.

SUBJECT: Requests approval for use of relief request number 36 concerning ASME Section Category B-F,to address surface examinations of identified Class 1 nozzle-to-safe end welds associated with reactor pressure vessel.

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RG AND E
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ROBERT C. MECREDY
Vice President
Nuclear Operations

December 18, 1998

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Inservice Inspection Program ASME Section XI
Required Examinations
Third 10-Year Interval
Request for Relief Regarding Request No. 36
R.E. Ginna Nuclear Power Plant
Docket No. 50/244

Reference: (a) Letter from W.R. Butler (NRC) to R.C. Mecredy (RG&E), dated September 8, 1993, Subject: Relief Request No. 19

Dear Mr. Vissing:

The purpose of this letter is to seek approval for the use of Relief Request number 36 concerning ASME Section XI Category B-F, to address surface examination limitations associated with weld examinations of identified Class 1 nozzle-to-safe end welds associated with the Reactor Pressure Vessel. 1/1

This Relief is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii), the required examination coverage for the identified welds is impractical and would require redesign or replacement to obtain Code required surface examination coverage. A047 Justification and the proposed alternative are included in the attachment to this letter. It is requested that this relief request be expedited, and NRC reply obtained before the end of January, 1999, in order for it to be utilized at R.E. Ginna Nuclear Power Plant for the upcoming March 1999 outage.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

.C. .19

Attachments: 3

xc: Mr. Guy S. Vissing (Mail Stop 14B2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

ATTACHMENT 1

Rochester Gas and Electric Corporation
Ginna Station
Docket No. 50/244
Third 10-Year Interval
Request for Relief No. 36
Reactor Pressure Vessel Nozzle-to-Safe End Butt Weld Surface Examination Limitations

I. System/Component(s) for Which Relief is Requested:

This Relief Request is requested for six (6) Reactor Pressure Vessel Nozzle-to-Safe End Butt Welds. Inspection of these welds is addressed under Class 1, Category B-F, Item Number B5.10, Nozzle-to-Safe End Weld Surface Examinations as identified below.

Weld ID #	ISI Summary #	Coverage Obtained
PL-FW-II	002100	74%
PL-FW-V	002400	76%
PL-FW-IV	002700	70%
PL-FW-VII	003000	76%
AC-1003-1	003300	0%(*)
AC-1002-1	003600	0%(*)

Note: (*) = welds embedded in concrete.

II. Code Requirement:

Under Category B-F, Item Number B5.10, volumetric and surface examinations shall be performed with essentially 100% of the weld length to obtain code coverage. ASME Section XI Code Case N-460 states that if the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in coverage is acceptable provided that the (lack of) coverage is less than 10%. Previous Codes utilized did not include this 90% coverage requirement and examinations were performed to the extent obtainable.

III. Code Requirement from Which Relief is Requested:

Relief is requested from the surface examination requirements for the six (6) identified welds.

Surface Examination of the first four (4) welds is limited due to Original Construction Code interferences of the floor and wall in the "Sandbox" where these welds are located. The "Sandboxes" would have to be redesigned to enable the welds to be surface examined to obtain Code required coverage. (Volumetric examination of these welds is performed from the inside of the Vessel and is not a part of this Relief Request.)

Surface Examination of the last two welds is impractical. The concrete surrounding the Reactor Pressure Vessel has embedded these welds. The concrete wall around the Reactor Pressure Vessel would have to be redesigned or replaced to enable the two (2) welds to be inspected with a surface examination. (Volumetric examination of these welds is performed from the inside of the Vessel and is not a part of this Relief Request.)

IV. Basis for Relief:

Relief is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii), the required examination coverage for the identified welds is impractical and would require redesign or replacement to obtain Code required surface examination coverage.

R.E. Ginna Nuclear Power Plant was designed and constructed to the B31.1, 1955 edition Construction Code. This code did not contain requirements to ensure that items be accessible for future examinations. The above noted piping welds were installed utilizing this construction code, which did not provide for accessibility for future ISI NDE. Due to the limited design accessibility, ISI surface examination coverage is below Code percentage requirements as identified within this Relief Request.

The first four (4) welds of this Relief Request are located in a "Sandbox" configuration. Within the "Sandbox", the welds are against the floor and one wall. The angled wall is joined to the floor and is against the weld. The surface examination of these welds is limited due to Original Construction Code interferences of the floor and wall of the "Sandbox". The "Sandboxes" would have to be redesigned to enable the welds to be inspected to obtain Code required coverage for the surface examinations. The attached sketch (Attachment 2) shows a representative weld with similar interferences.

The last two (2) welds of this Relief Request are embedded in concrete. This concrete structure is the wall that surrounds the Reactor Pressure Vessel.

ASME Section XI Class 1 system leakage examinations are performed. These leakage examinations demonstrate pressure boundary integrity and provide additional assurances in maintaining plant safety.

V. Alternate Examinations:

R.E. Ginna Nuclear Power Plant proposes that the surface examination coverage identified for the first four (4) welds above be acceptable in fulfilling the Code required examination coverage. The actual physical configuration of the "Sandboxes" is not conducive in obtaining the requirements specified within Code Case N-460 for acceptable coverage. Volumetric examination of these welds is performed from the inside of the Vessel, and will be performed during the 1999 outage.

For the last two (2) welds, the Code surface examination requirements are impractical and cannot be examined due to them being embedded in concrete. Volumetric examination of these welds is performed from the inside of the Vessel, and will be performed during the 1999 Outage.

VI. Justification for the Granting of Relief:

R.E. Ginna Nuclear Power Plant was designed and constructed to the B31.1, 1955 edition Construction Code. This code did not contain requirements to ensure that items be made accessible for future NDE examinations. Due to the original limited design accessibility or lack of design accessibility, ISI surface examination coverage can not be obtained to the extent required by the ASME Code.

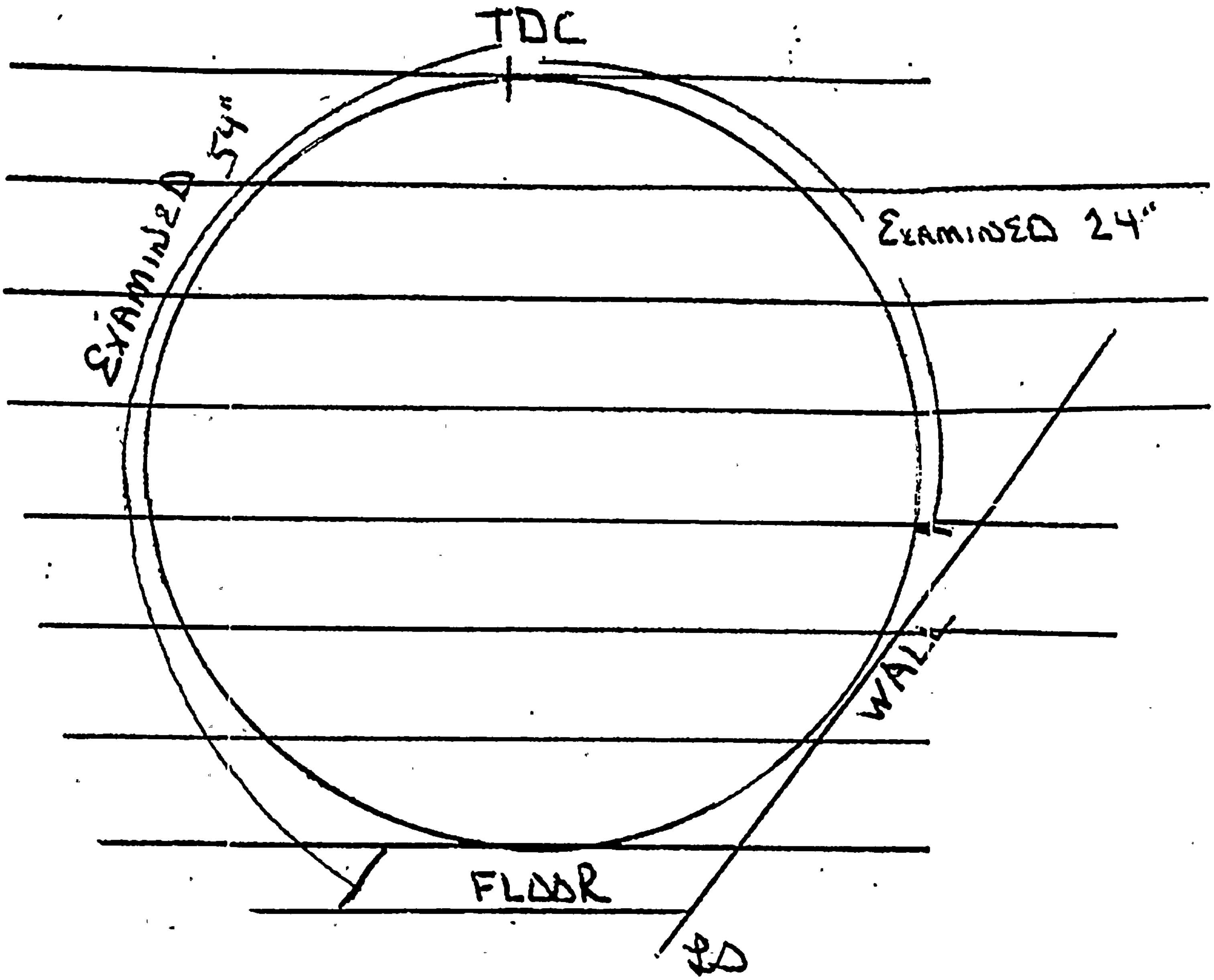
ASME Section XI Class 1 system leakage examinations are performed. These leakage examinations demonstrate pressure boundary integrity and provide additional assurances in maintaining plant safety. The identified examination coverage for these items should be acceptable in fulfilling ASME Section XI coverage requirements.

It should also be noted that Relief Request Number 36 is similar to RG&E's Relief Request Number 19, for which relief was previously granted. See Attachment 3.

VII. Implementation Schedule:

The surface examinations will be performed during the 1999 Outage on the first four (4) accessible welds. For the two (2) welds embedded in concrete, surface examinations can not be performed. The associated volumetric examinations will be performed during the 1999 Outage. Applicable Code credit shall be taken for the Third 10-year Interval inspection, upon approval of this Relief Request.

Attachment 2



EXAMINATION AREA LIMITATION (IF NONE, SO STATE):
Due to Configuration of Floor and wall no Exam in Area not Identified in Drawing

REVIEWED BY:	[illegible] BY	SNT LEVEL:	DATE:
R. L. May	[illegible signature] 4/15/89	II	15-APR-89

FORM NO. Sw.R.I NDTR 17-11 (REV. 1-3-79)

Attachment 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

G. Wrobel to prepare response. Copies for S Adams, M Saporito, G Voci

September 8, 1993

Docket No. 50-244

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Dr. Mecredy:

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT - THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN AND ASSOCIATED REQUESTS FOR RELIEF (TAC NOS. M84044 AND M86225)

RG014471

By letter dated July 21, 1989, you submitted the R. E. Ginna Nuclear Power Plant Third 10-Year Interval Inservice Inspection Program Plan. The NRC concluded in its safety evaluation (SE) dated August 6, 1990, that the Program Plan, with the exception of Requests for Relief Nos. 10 and 13, was found to be acceptable and in compliance with the regulations.

Subsequent to the NRC's review above, you submitted two revisions of the Program Plan. Revision 1 was submitted in a letter dated August 10, 1992, and Revision 2 was submitted by letter dated January 25, 1993. The January 25, 1993, submittal of the Program Plan was reformatted for ease of use. The reformatted program consisted of eleven independent sections, each carrying its own revision number.

In your letter dated January 5, 1993, you submitted Relief Request (RR) No. 19 and notified the NRC of the intent to incorporate Code Cases N-460 and N-498 into the Program Plan.

The NRC staff, with the assistance of its contractor, Idaho National Engineering Laboratory (INEL), reviewed and evaluated your submittal dated January 5, 1993, and concluded that pursuant to 10 CFR 55.55a(g)(6)(i), relief can be granted as requested for RR No. 19.

Regarding Revision 2 and the January 25, 1993, submittal of the Program Plan, the staff has concluded that your response regarding the removal of insulation during pressure testing at bolted connections in piping systems used for controlling boration is still considered unacceptable, and you should either withdraw Request for Relief No. 13 or acknowledge it as being unacceptable by the staff in the Program Plan.

Therefore, due to inadequate VT-2 visual examinations of the bolted connections in borated systems, and Request for Relief No. 13 not having been

withdrawn, the staff concludes that the Revision 2 submittal of January 25, 1993, is not in compliance with 10 CFR 50.55a(g) and Technical Specification 4.2.1.5 and is therefore unacceptable. The staff's evaluation and conclusions are contained in the attached SE.

Sincerely,

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

cc:

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Enclosure

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

REQUESTS FOR RELIEF

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NUMBER 50-244

1.0 INTRODUCTION

Technical Specification 4.2.1.5 for the R. E. Ginna Nuclear Power Plant states that the inservice inspection and testing of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The Code of Federal Regulations of 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during each 10-year interval comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein. The applicable Edition of Section XI of the ASME Code for the R. E. Ginna Nuclear Power Plant Third 10-Year Interval is the 1986 Edition, no addenda. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

By letter dated July 21, 1989, Rochester Gas and Electric Corporation (the licensee) submitted the R. E. Ginna Nuclear Power Plant Third 10-Year Interval Inservice Inspection Program Plan. In a Safety Evaluation Report (SER) dated August 6, 1990, the staff found the Program Plan, with the exception of Requests for Relief Nos. 10 and 13, acceptable and in compliance with the regulations.

Revision 1 of the R. E. Ginna Nuclear Power Plant Third 10-Year Interval Inservice Inspection Program Plan was submitted in a letter dated August 10, 1992, and a subsequent revision was submitted by letter dated January 25, 1993. The January 25, 1993, submittal of the Program Plan was reported to have been reformatted for ease of use, and supersedes the previous submittals in their entirety. The reformatted Program consists of eleven (11) independent sections, each of which carries its own revision number and may be revised separately.

In a letter dated January 5, 1993, the licensee submitted Relief Request (RR) No. 19 and notified the NRC of the intent to incorporate Code Cases N-460 and N-498 into the Program Plan. These items are also addressed in the following section.

The staff, with technical assistance from its Contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the R. E. Ginna Nuclear Power Plant Third 10-Year Interval Inservice Inspection Program Plan, as submitted January 25, 1993, and the January 5, 1993, submittal which includes RR No. 19. The results are reported below.

2.0 EVALUATION

The following are the major changes that have been incorporated into the January 25, 1993, revision of the R. E. Ginna Nuclear Power Plant Third 10-Year Interval Inservice Inspection Program Plan:

(a) Relief Request No. 4 (Reactor Coolant Pump Casing Welds and Internals): Based on the Licensee's use of ASME Code Case N-481, "Alternative Examination Requirements for Cast Austenitic Pump Casings, Section XI, Division 1," RR No. 4 is no longer required and was withdrawn in Section 2 of the revised Program Plan. Code Case N-481 is acceptable for general usage as it is referenced in NRC Regulatory Guide 1.147, Revision 9, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1."

(b) Licensee's use of an Authorized Nuclear Inservice Inspector (ANII): In a letter, A. Johnson (NRC) to Dr. R. C. Mecredy (RG&E), dated June 16, 1992, the NRC requested that the licensee confirm that all duties were being performed by an ANII as required by the Code. In the response dated August 17, 1992, [Dr. R. C. Mecredy (RG&E) to Document Control Desk (NRC)], the licensee committed to contract with the

Hartford Steam Boiler Inspection and Insurance Company for services of an ANII for the Third 10-Year Inspection Interval. The licensee stated that the ANII will perform all required Code duties in accordance with IWA-2110. Consequently, the licensee withdrew RR No. 3 (Use of an Authorized Inspection Agency to Provide Inspection Services) in Section 2 of the revised Program Plan.

(c) NIS-1 and NIS-2 Forms: The NRC's June 16, 1992, letter to the licensee also addressed the use of NIS-1 (Owner's Report for Inservice Inspections) and NIS-2 (Owner's Report for Repairs or Replacements) forms. These forms are specified in Mandatory Appendix II of ASME Code Section XI. IWA-6220(d)(10) states that the NIS-1 and NIS-2 forms shall be included in the required Inservice Inspection Summary Report and that they include the signature of the ANII. Therefore, Section 1.6.1 of the Program Plan, in the latest revision, has been revised to include use of the NIS-1 and NIS-2 forms. Section 1.6.1 now states:

> "An Inservice Inspection Report shall be generated to document applicable inservice inspection and associated repair, replacement and modification activities. ASME NIS-1 and NIS-2 forms shall be generated and included within the Inservice Inspection Report."

(d) Removal of insulation at bolted joints in piping systems for controlling boration during pressure testing: In the June 16, 1992, letter to the licensee, the staff did not agree with the licensee's basis for limiting the extent of removal of insulation to inspections at bolted connections with ferrous steel fasteners. A non-isolatable leak could occur anywhere in the piping systems used for controlling boration regardless of fastener material types. Therefore, the licensee was requested to satisfy the Code requirements regarding VT-2 visual examinations at bolted connections.

In the response dated August 17, 1992, the licensee agreed with the staff's evaluation and stated that paragraph 1.10.3.2 would be revised to require the removal of insulation for inspection of both ferritic and austenitic bolting. Section 1.10.3.2 of the January 25, 1993, Program Plan submittal was revised to state, in part, that:

> "Insulation removal during the VT-2 examination is not required, however, in accordance with IWA-5242(a), systems borated for the purpose of controlling reactivity shall have insulation removed at bolted connections during conduct of the VT-2 examination. This requirement is only applicable to those VT-2 examinations performed during a hydrostatic test, since Leakage, Functional and Inservice tests are intended to be non-intrusive type tests. At Ginna, this requirement is considered to be applicable to borated lines only in the primary flow path of piping from the boric acid supply and CVCS Charging to the Reactor Vessel and return through CVCS Letdown, and

is not applicable to branch lines connected to the primary flow path."

The staff considers this response unacceptable. The licensee is not intending to perform any hydrostatic testing based on the use of ASME Code Case N-498, Alternative Rules for 10-Year Hydrostatic Pressure Testing for Class 1 and 2 Systems, Section XI, Division 1. Additionally, Section XI of the Code requires the removal of insulation at bolted connections on all systems that contain borated water during the conduct of a VT-2 visual examination. This does not exclude VT-2 visual examinations during functional or inservice tests. ASME Code Interpretation XI-1-89-38 supports this conclusion and should be referenced if further clarification of this requirement is necessary. For the January 25, 1993, revision of the Program Plan to be considered acceptable, the licensee must meet the Code requirements regarding bolted connections on systems containing borated water.

(e) As stated in Section 1.0 of this report, the staff denied Requests for Relief Nos. 10 and 13 in the SER dated August 5, 1990. It is noted in the January 25, 1993, revision of the Program Plan that Request for Relief No. 10 has been withdrawn. However, Request for Relief No. 13 has not been withdrawn and appears to be applicable for the current 10-year inspection interval. Request for Relief No. 13 should either be withdrawn, or acknowledged in the Program Plan as being NRC unacceptable.

The following evaluations address the January 5, 1993, letter notifying the staff of the licensee's intent to incorporate Code Cases N-460 and N-498 and the submittal of RR No. 19.

(f) ASME Code Cases N-460 and N-498 have both been approved for use by reference in Regulatory Guide 1.147, Revision 9, "Inservice Inspection Code Case Acceptability ASME Section XI Division 1", dated April 1992.

(g) <u>Relief Request No. 19, Examination Category C-B, Items C2.21 and C2.22, Charging System Pulsation Dampener Nozzle Welds and Inside Radius Sections</u>

<u>Code Requirement</u>: Section XI, Table IWC-2500-1, Examination Category C-B, Item C2.21 requires a 100% surface and volumetric examination of nozzle-to-shell welds on nozzles in vessels with nominal wall thickness ≥1/2 inch. Item C2.22 requires a 100% volumetric examination of the inside radius sections of the nozzles. These examinations are to be performed as defined by Figures IWC-2500-4(a) or (b) as applicable.

Licensee's Code RR: Relief is requested from performing the surface and volumetric examinations to the extent required by the Code for the following charging system pulsation dampener nozzle welds and inside radius sections:

Nozzle	NDE Method	Coverage
CF-N1	PT	66%
	UT	65%

Nozzle	NDE Method	Coverage
CF-N2	PT	66%
	UT	65%
CF-N3	PT	>90%
	UT	80%

Licensee's Basis for Requesting Relief: The pulsation dampener contains three (3) nozzles, in line, located at the bottom of the unit. The outboard nozzle is identified as CF-N1. Between this nozzle and the middle nozzle (CF-N2) is a support that covers from the edge of one nozzle's weld heat affected zone to the edge of the other nozzle's weld heat affected zone. There is only 7/8 inch between CF-N2 and the third nozzle (CF-N3) heat affected zone.

Licensee's Proposed Alternative Examination: None. The Code-required surface and/or volumetric examinations will be performed to the maximum extent practical.

Evaluation: The R. E. Ginna Nuclear Power Plant was constructed to the 1955 Edition of ANSI B31.1. This Code did not contain requirements to ensure that items be accessible for future examinations. The pulsation dampener was constructed and installed in the early 1970s, and the construction code did not require provisions for accessibility for inservice inspections. Due to the close proximity of the nozzles and/or the vessel support, the associated surface and volumetric examinations are impractical to perform to the extent require by the Code. The identified surface and volumetric examination coverage of 66% to >90% should be considered acceptable for these nozzles at R. E. Ginna Nuclear Power Plant.

Conclusion: Based on the above evaluation, it is concluded that since the original construction code did not specify accessibility requirements for future ISI NDE, compliance with the Code for these nozzle examinations is impractical. Imposition of the surface and volumetric examinations, to the extent required by the Code, would necessitate redesign or replacement of the charging system pulsation dampener and result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), relief is granted as requested.

3.0 CONCLUSION

Paragraph 10 CFR 50.55a(g)(4) requires that components (including supports) that are classified as ASME Code Class 1, 2, and 3 meet the requirements, except design and access provisions and preservice requirements, set forth in applicable Editions of ASME Section XI to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee determined that conformance with certain Code requirements is impractical for their facility and submitted supporting technical justification. The staff has reviewed the licensee's submittal, dated January 5, 1993, and has concluded that pursuant to 10 CFR 55.55a(g)(6)(i) relief can be granted as requested for RR No. 19. Such relief is authorized by law and will not endanger life, property, or the common defense and security, and is otherwise in public interest. This relief is being granted giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Regarding Revision 2 and the January 25, 1993, submittal of the Program Plan, the staff has concluded that the licensee has adequately addressed the deficiencies cited in the June 16, 1992, letter from the NRC regarding the licensee's use of an ANII and the NIS-1 and NIS-2 forms. However, as addressed above, the licensee's response regarding the removal of insulation, during pressure testing, at bolted connections in piping systems used for controlling boration is still considered unacceptable. In addition, the licensee should either withdraw Request for Relief No. 13 or acknowledge it as being unacceptable to the NRC in the Program Plan.

Based on inadequate VT-2 visual examinations of bolted connections in borated systems, and Request for Relief No. 13 not having been withdrawn, the staff concludes that the R. E. Ginna Nuclear Power Plant Third 10-Year Interval Inservice Inspection Program Plan, Revision 2, as submitted January 25, 1993, is not in compliance with 10 CFR 50.55a(g) and Technical Specification 4.2.1.5 and is therefore unacceptable.

Principal Contributors: T. McLellan
M. Khanna

Date: